UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 18, 2015

Mr. Michael P. Gallagher
Vice President, License Renewal Projects
Exelon Generation Company, LLC
200 Exelon Way
Kennett Square, PA 19348

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE LASALLE COUNTY STATION, UNITS 1 AND 2 LICENSE RENEWAL APPLICATION – SET 9 (TAC NOS. MF5347 AND MF5346)

Dear Mr. Gallagher:

By letter dated December 9, 2014, Exelon Generation Company, LLC (Exelon) submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, to renew the operating licenses NPF-11 and NPF-18 for LaSalle County Station (LSCS), Units 1 and 2, respectively. The staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review.

These requests for additional information were discussed with Mr. John Hufnagel, and a mutually agreeable date for the response is within 30 days from the date of this letter. If you have any questions, please contact me at 301-415-3019 or by e-mail at Jeffrey.Mitchell2@nrc.gov.

Sincerely,

/RA/

Jeffrey S. Mitchell, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket Nos. 50-373 and 50-374

Enclosure:
As stated

cc: Listserv

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Letter to Michael P. Gallagher from Jeffrey S. Mitchell dated August 18, 2015

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LASALLE COUNTY STATION, UNITS 1 AND 2
LICENSE RENEWAL APPLICATION
REQUESTS FOR ADDITIONAL INFORMATION – SET 9
(TAC NOS. MF5347 AND MF5346)

RAI B.2.1.7-3a

Background:

By letter dated June 25, 2015, the applicant responded to request for additional information (RAI) B.2.1.7-3 that addressed limited examination coverage for Generic Letter (GL) 88-01 welds. In its response, the applicant assessed a time period (February 2008 through March 2015) to evaluate the limited examination coverage of the welds within the scope of the BWR Stress Corrosion Cracking program. The applicant indicated that during the assessed period (February 2008 through March 2015) none of the Unit 1 Category A welds and Unit 2 Category C welds were inspected.

The applicant also indicated that there are only two Category D welds at Unit 2. The applicant further indicated that the examination coverage for each of the welds was limited to 50 percent because each weld is between pipe and a cast valve body where the current approved Performance Demonstration Initiative (PDI) method does not provide qualified and reliable results for the side of the weld adjacent to the cast valve body due to the specific contour of the valve body.

Issue:

As described in the background section, the response did not provide information regarding the examination coverage of Unit 1 Category A welds and Unit 2 Category C welds. The staff needs information on examination coverage of these welds based on previous inspection results in order to resolve the concern related to limited examination coverage.

The staff also needs additional information to evaluate the applicant's justification for the limited examination coverage of the Unit 2 Category D welds. It is unclear whether the cast valve bodies associated with the Unit 2 Category D welds are made of cast materials resistant to intergranular stress corrosion cracking (IGSCC). In addition, the staff needs to confirm whether inspection results for the cast valve bodies indicate occurrence of IGSCC.

Request:

1. Provide information on the average examination coverage of Unit 1 Category A welds and Unit 2 Category C welds to characterize the overall degree of limited examination coverage. If limited examination coverage is identified for these welds, provide justification for why additional inspections are not necessary to compensate for the limited examination coverage of these welds.

ENCLOSURE

2. Provide the following information regarding the cast valve bodies associated with the two Category D welds at Unit 2:

    a) Cast material grade (e.g., CF3M), carbon content, and ferrite content (including the method for determining ferrite content) of the cast valve bodies to clarify whether the materials are resistant to IGSCC in accordance with the staff positions in GL 88-01

    b) American Society of Mechanical Engineers (ASME) Code Class and, if available, examination results of the cast valve bodies such as internal visual or surface examination results per ASME Code Section XI requirements

RAI 4.2.1-1

Background:

License Renewal Application (LRA) Section 4.2.1 describes the applicant's time-limited aging analysis (TLAA) on reactor vessel fluence calculations. LRA Section 4.2.1 also indicates that the 54-effective full power year (EFPY) fluence projections for 60 years of operation were calculated by using the NRC-approved Radiation Analysis Modeling Application (RAMA) methodology. The LRA further states that the 54-EFPY fluence projections compile the cumulative fluence resulting from each past operating cycle and add the predicted fluence estimate for future operating cycles through the period of extended operation. As discussed below, the NRC staff noted that these 54-EFPY fluence projections are independent from the fluence projections used in the current P-T limits.

LRA Section 4.2.1 states that the neutron fluence projections used as inputs to the current 40-year neutron embrittlement analyses for LSCS, Units 1 and 2 were developed in accordance with GE Licensing Topical Report NEDO-32983, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations," (ADAMS Accession No. ML072480121). LRA Section 4.2.1 also indicated that this GE methodology is in compliance with NRC Regulatory Guide 1.190, as approved by the NRC staff in the safety evaluation dated September 14, 2001.

Under LRA Section 4.2.1, the TLAA evaluation states that the 60-year RAMA neutron fluence projections compile the cumulative neutron fluence resulting from each past operating cycle and add the predicted neutron fluence estimated for future operating cycles through the period of extended operation. The RAMA neutron fluence projections are prepared using the BWRVIP-126 methodology. These 60-year neutron fluence projections are independent from the neutron fluence projections used with the current P-T curve submittals.

In comparison, the following reference indicates that the current 40-year (32-EFPY) P-T limits for LSCS Unit 1 use neutron fluence values that are calculated via the RAMA fluence methodology.

- NRC Safety Evaluation, "LaSalle County Station, Unit 1, Issuance of Amendment Revising Pressure and Temperature Limits (TAC No. MF3270)," dated November 25, 2014 (ADAMS Accession No. ML14220A517).

Section 3.1, Neutron Fluence Calculation, of the above reference states that the safety evaluation shall not be constructed as endorsement, agreement with, or approval of, the position regarding the combined use of neutron fluence methods to determine a total neutron fluence.

Issue:

The LRA does not address the RAMA methodology that the NRC staff evaluated in the safety evaluation regarding the current P-T limits for LSCS, Unit 1. Therefore, the NRC staff needs additional clarification as to the specific neutron fluence methodology that is used in the license renewal TLAAs. It is also unclear to the NRC staff how the applicant evaluates potential effects of updated neutron fluence calculation on existing neutron embrittlement analysis.

Request:

1. Clarify whether the 54-EFPY neutron fluences described in the LRA are based on the GE methodology, the RAMA methodology, or the combination of the two methodologies.
2. Clarify how the applicant will ensure that the actual fluence levels are bounded by the fluence levels analyzed in LRA Section 4.2.1.

RAI 4.2.8-1

Background:

LRA Section 4.2.8 describes the TLAA for the loss of preload of the reactor pressure vessel (RPV) core plate rim hold-down bolts resulting from irradiation effects. The LRA states that a fluence evaluation based on 54 EFPY has identified the bolts with the highest fluence values after 60 years of operation for Unit 1 (3.60×10^{19} n/cm^2) and Unit 2 (3.85×10^{19} n/cm^2). The LRA also states that an average fluence value of 8.0×10^{19} n/cm^2 was evaluated for 40 years of operation which resulted in a maximum relaxation of 19 percent in preload for the RPV core plate rim hold-down bolts. The TLAA is dispositioned in accordance with 10 CFR 54.21(c)(1)(i) to remain valid through the period of extended operation.

The loss of preload analysis was performed as part of Boiling Water Reactor Vessel and Internals Project Technical Report 25 (BWRVIP-25), "BWR Core Plate Inspection and Flaw Evaluation Guidelines." Section 2.1.3 of BWRVIP-25 states that it has been determined that a 5 percent to 19 percent reduction in preload is expected over a 40-year operating experience and that the bolts will maintain some amount of preload throughout the life of the plant. Section B.4 of Appendix B states that the loss of preload was recalculated based on 60 years of plant operation and that the reduction in preload remains 5 percent to 19 percent.

Issue:

The staff is unable to determine if the amount of relaxation experienced after 60 years of operation is bounded by the 40-year analyses because the fluence values used to determine the range in percent reduction of preload are not provided in BWRVIP-25 or the appendixes of the report.

Request:

Provide the analysis used to determine that an average fluence value of $8.0x10^{19}$ n/cm^2 produces a maximum relaxation of 19 percent in preload for the RPV core plate rim hold-down bolts. Justify that the analysis adequately considers plant-specific configurations of reactor vessel internals and is bounded by the relaxation evaluations in BWRVIP-25.

RAI 4.2.10-1

Background:

Section 4.7.3.1.3 of the SRP-LR (NUREG-1800, Rev. 2) provides the NRC's review procedures for plant-specific TLAAs which will be managed in accordance with 10 CFR 54.21(c)(1)(iii). SRP-LR Section 4.7.3.1.3 states that the staff is to review the aging management program proposed by the applicant to verify the program is adequate to manage the aging effects associated with the TLAA.

LRA Section 4.2.10 describes the TLAA evaluation for the loss of preload of the jet pump slip joint repair clamp. LRA Section 4.2.10 dispositions the TLAA in accordance with 10 CFR 54.21(c)(1)(iii) to be managed by Commitment No. 47 in LRA Section A.5. This commitment states that:

> Prior to exceeding the limiting fluence value of $1.17x10^{20}$ n/cm^2 at the Unit 1 jet pump slip joint clamp location, estimated to be at 50.7 EFPY, revise the analysis for the slip joint clamps for a higher acceptable fluence value or take other corrective action such as repair or replacement of the clamps to ensure acceptable clamp preload.

The implementation schedule for Commitment No. 47 is "Prior to the period of extended operation."

Issue:

The LRA does not clearly identify how the applicant will ensure that the limiting fluence value is not exceeded. The staff is unable to determine if the applicant's program or activities are adequate to ensure that the limiting fluence value is not exceeded.

Request:

Identify the program or a set of activities that will be used for the jet pump slip joint repair clamp to ensure that the limiting fluence value is not exceeded. Justify that the program or activities are adequate for managing the aging effect of loss of preload of the jet pump slip joint repair clamp.